UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 9, 1984

Docket No. 50-249
LS05-84-03-013

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

SUBJECT: TECHNICAL SPECIFICATION CHANGES RELATING TO THE CYCLE 9 RELOAD FOR DRESDEN NUCLEAR POWER STATION, UNIT NO. 3

The Commission has issued the enclosed Amendment No. 74 to Facility Operating License No. DPR-25 for the Dresden Nuclear Power Station, Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application dated July 18, 1983 and August 25, 1983 as supplemented by letters dated November 3, 10, and 30, 1983, two letters dated December 13, 1983, and a letter dated December 16, 1983.

The amendment authorizes changes to the Technical Specifications to support Cycle 9 operation of Dresden 3 with reload fuel supplied by and the associated analyses performed by the Exxon Nuclear Company. The amendment also authorizes Dresden 3 to install eight lead control blades designed and built by ASEA-Atom. Specifically related to the operation with an Exxon fuel reload, the amendment authorizes (1) a revision of the MAPLHGR curves for Dresden Unit 3, Cycle 9, (2) replacement of the K_f curve with Exxon Nuclear Corporation's reduced flow MCPR limits and (3) an administrative change to the bases of the reactor coolant safety limit specification which corrects an oversight in the Dresden Unit 3, Cycle 8 submittal.

Notices of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the actions requested in the July 18 and August 25, 1983 letters were published in the Federal Register on November 22, 1983 (48 FR 52807 and 52808). No request for hearing was received. A verbal comment from Mr. R. Minue of the Illinois Department of Nuclear Safety was received on December 5, 1983. His concern was related to the indications of cracking in ASEA-Atom control blades at high burnup as discussed in the licensee's November 10, 1983 supplemental letter. The blade cracking issue is addressed in Section 2.5 of the staff's safety evaluation. The supplementary letters furnished clarifying information needed by the staff but made no changes in the content of the amendments and were, therefore, encompassed within the prenotices published November 22, 1983.

SE01
DSU USE 16

A copy of our related Safety Evaluation is also enclosed. This action will appear in the Commission's Monthly Notice Publication in the Federal Register.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 74 to DPR-25
2. Safety Evaluation

cc w/enclosures:
See next page

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Dennis M. Crutchfield, Chief
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Mr. Dennis L. Farrar

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Chairman
Board of Supervisors of
Grundy County
Grundy County Courthouse
Morris, Illinois 60450

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Federal Activities Branch
Region V Office
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Chicago, Illinois 60604

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Nuclear Facility Safety
Illinois Department of Nuclear Safety
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
License No. DPR-25

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by the Commonwealth Edison Company (the licensee) dated July 18, 1983, and August 25, 1983, as supplemented by letters dated November 3, 10 and 30, 1983, two letters dated December 13, 1983, and a letter dated December 16, 1983 comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-25 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 9, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 74

FACILITY OPERATING LICENSE DPR-25

DOCKET NO. 50-249

Revise the Technical Specifications by replacing the following pages with attached revised pages. These revised pages contain the captioned amendment number and marginal lines to reflect the area of change.

Remove Pages	Insert Page
20	20
81C-1	81C-1
81D	81D
81E	81E
--	81E-1
86A	86A
157	157

Bases:

1.2 The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1345 psig as measured by the vessel steam space pressure indicator ensures margin to 1375 psig at the lowest elevation of the reactor vessel. The 1375 psig value is derived from the design pressures of the reactor pressure vessel and coolant system piping. The respective design pressures are 1250 psig at 575°F and 1175 psig at 560°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and USASI 831.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% X 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% X 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the reactor vessel. The design pressure for the recirc. suction line piping (1175 psig) was chosen relative to the reactor vessel design pressure. Demonstrating compliance of the peak vessel pressure with the ASME overpressure protection limit (1375 psig) assures compliance of the suction piping with the USASI limit (1410 psig). Evaluation methodology used to assure that this safety limit pressure is not exceeded for any reload as documented in Reference XN-NF-79-71. The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig: this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At that pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationships of stress levels to yield strength are comparable for the primary system piping and provide a similar margin of protection at the established safety pressure limit.

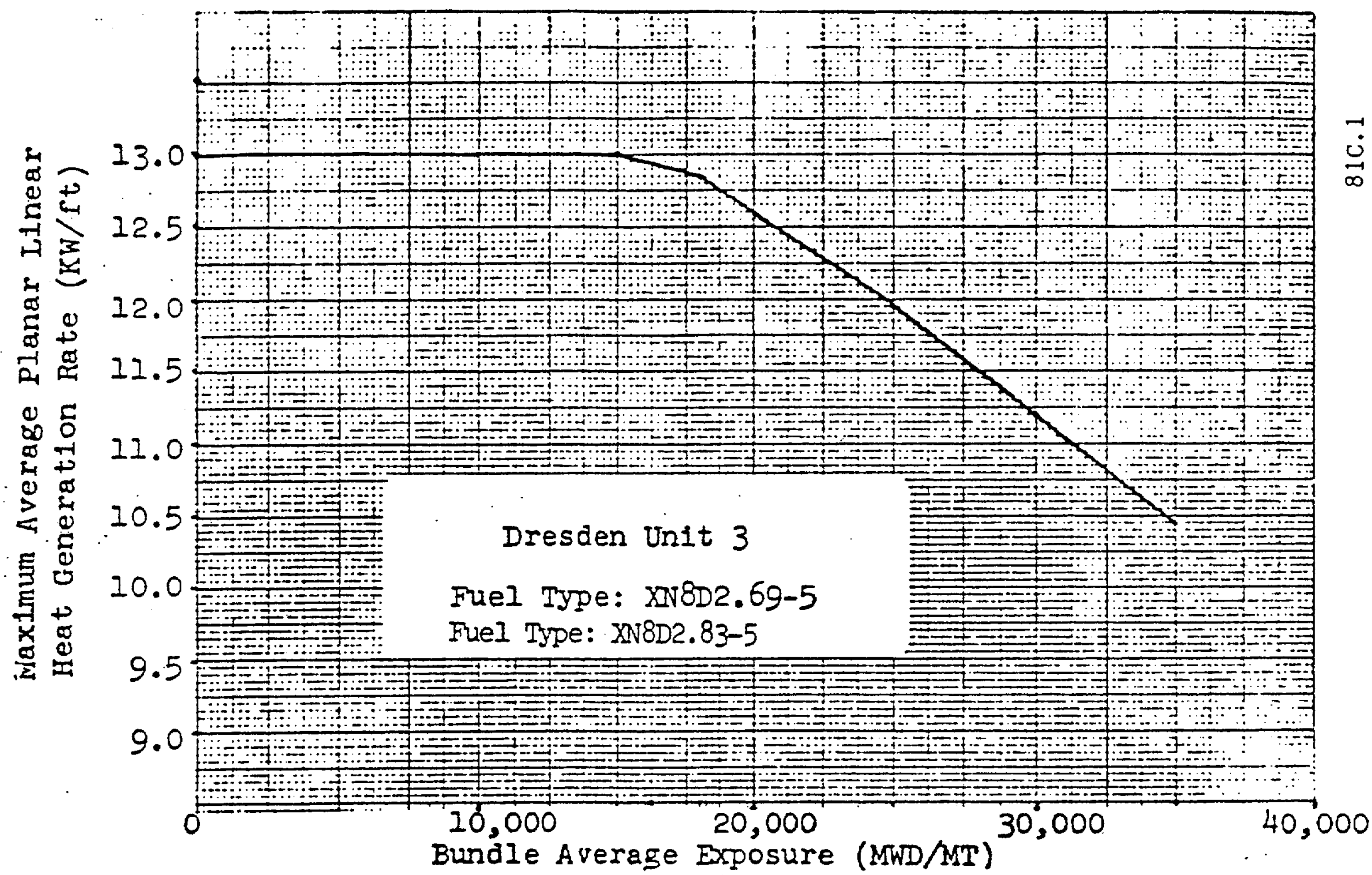
The normal operating pressure of the reactor coolant system is 1000psig. For the turbine trip or loss of electrical load transients, the turbine trip scram or generator load rejection scram, together with the turbine bypass system, limit the pressure to approximately 1100psig (2). In addition, pressure relief valves have been provided to reduce the probability of the safety valves, which discharged to the drywell, operating in the event that the turbine bypass should fail.

Finally, the safety valves are sized to keep the reactor vessel peak pressure below 1375 psig with no credit taken for the relief valves during the postulated full closure of all MSIV's without direct (valve position switch) scram. Credit is taken for the neutron flux scram, however,

The indirect flux scram and safety valve actuation provide adequate margin below the peak allowable vessel pressure of 1375 psig.

Reactor pressure is continuously monitored in the control room during operation on a 1500 psi full scale pressure recorder.

(4) SAR, Section II.2.2 -
also: "Dresden 3 Second Reload License Submittal," 9-14-73
also: "Dresden Station Special Report No. 29 Supplement B."



81C.1

Figure 3.5-1
(Sheet 1 of 5)

Maximum Average Planar Linear
Heat Generation Rate (MAPLHGR)
vs. Bundle Average Exposure

3.5 LIMITING CONDITIONS FOR OPERATION

K. Minimum Critical Power Ratio (MCPR)

During steady state operation at rated core flow, MCPR shall be greater than or equal to -

1.34 for GE 8 x 8R fuel

1.33 for ENC and GE 8 x 8 fuel

For core flows other than rated, the MCPR Operating Limit shall be as follows:

1. Manual Flow Control - the MCPR Operating Limit shall be the value from Figure 3.5-2 sheet 1 or the above rated flow value, whichever is greater.
2. Automatic Flow Control - the MCPR Operating Limit shall be the value from Figure 3.5-2 sheet 1, sheet 2, or the above rated flow value, whichever is greatest.

If at any time during steady state power operation, it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

In the event the average 90% scram insertion time determined by Spec. 3.3.C for all operable control rods exceeds 2.58 seconds, the MCPR limit shall be increased by the amount equal to [0.0544T - 0.14] where T equals the average 90% scram insertion time for the most recent half-core or full core surveillance data from Spec. 4.3.C.

4.5 SURVEILLANCE REQUIREMENTS

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during a reactor power operation at ≥25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

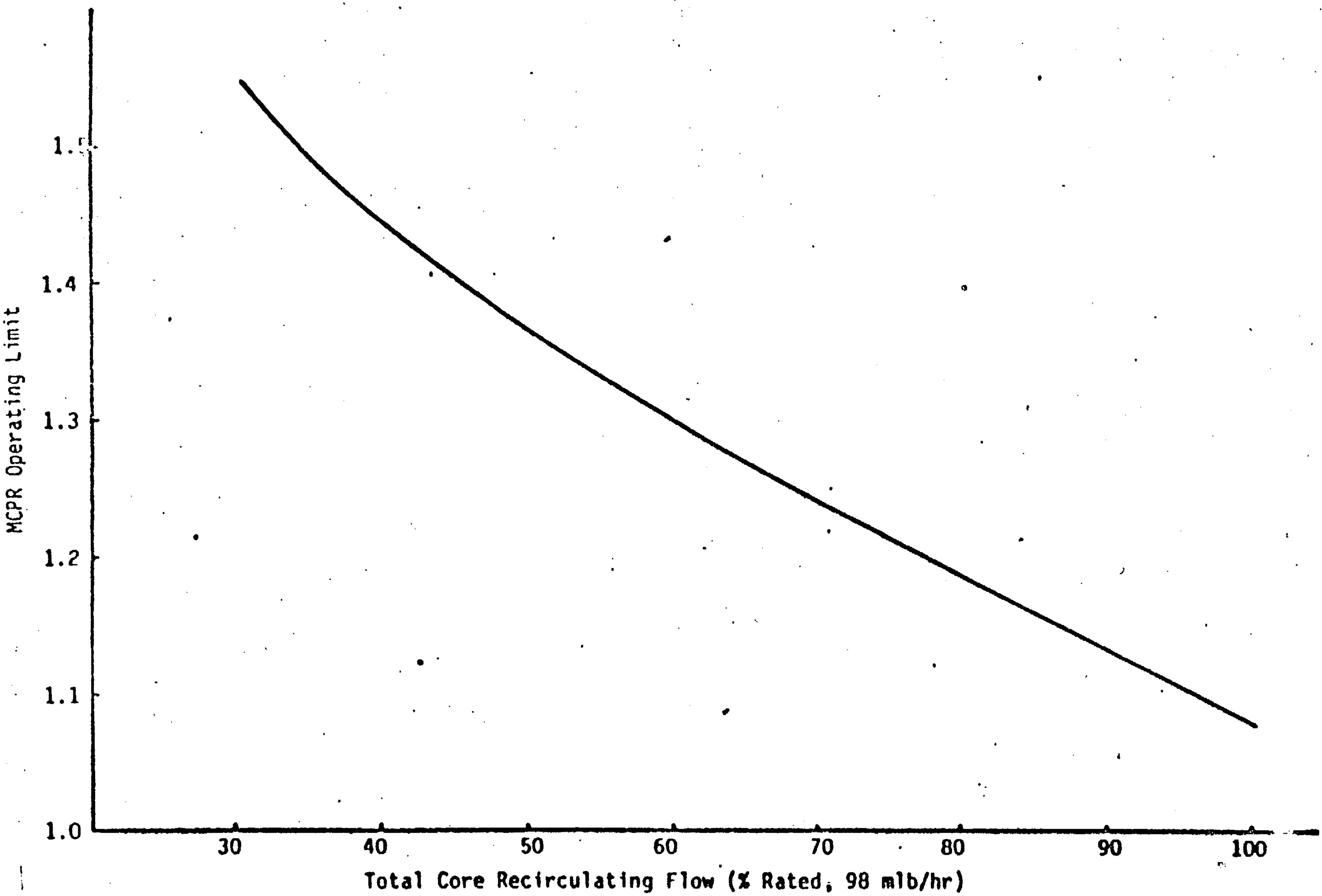


Figure 3.5-2 (Sheet 1 of 2) MCPR Limit For Reduced Core Flow

Amendment No. ~~5~~, 74

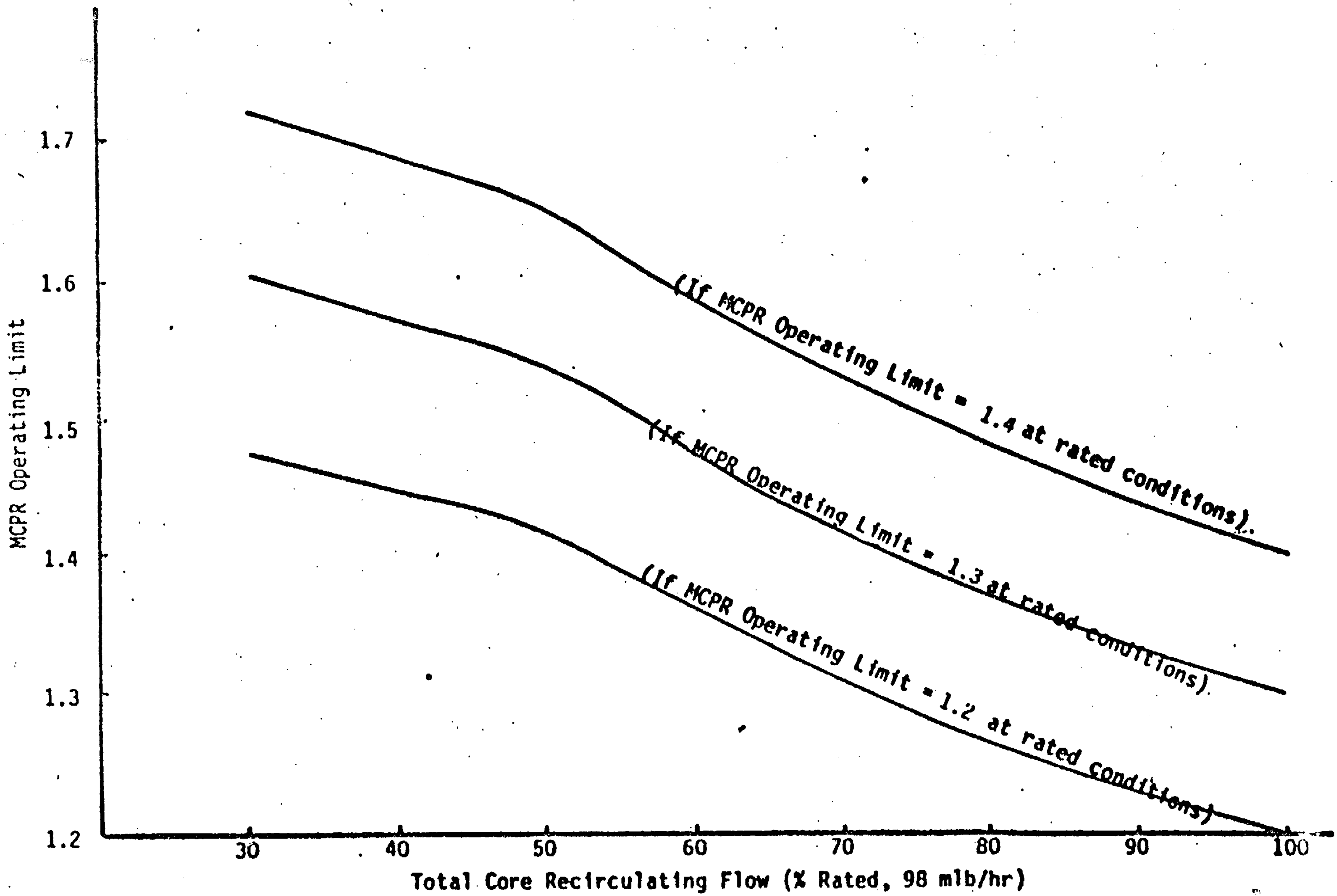


Figure 3.5-2 (Sheet 2 of 2) MCPR Limit For Automatic Flow Control

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

I. Average Planar LHGR

At core thermal power levels less than or equal to 25 per cent, operating plant experience and thermal hydraulic analyses indicate that the resulting average planar LHGR is below the maximum average planar LHGR by a considerable margin; therefore, evaluation of the average planar LHGR below this power level is not necessary. The daily requirement for calculating average planar LHGR above 25 per cent rated thermal power is sufficient since power distribution shifts are slow when there have not been significant power or control rod changes.

J. Local LHGR

The LHGR for G.E. fuel shall be checked daily during reactor operation at greater than or equal to 25 per cent power to determine if fuel burnup or control rod movement has caused changes in power distribution. A limiting LHGR value is precluded by a considerable margin when employing a permissible control rod pattern below 25% rated thermal power.

K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 per cent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point operating plant experience and thermal hydraulic analysis indicates that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR.

The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

5.0 DESIGN FEATURES

5.1 Site

Dresden Unit 3 is located at the Dresden Nuclear Power Station which consists of a tract of land of approximately 953 acres located in the northeast quarter of the Morris 15-minute quadrangle (as designated by the United States Geological Survey), Goose Lake Township, Grundy County, IL. The tract is situated in portions of Sections 25, 26, 27, 34, 35 and 36 of of Township 34 North, Range 8 East of the Third Principal Meridian.

5.2 Reactor

A. The core shall consist of not more than 724 fuel assemblies.

B. The reactor core shall contain 177 cruciformshaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% of theoretical density, or Hafnium metal.

5.3 Reactor Vessel

The reactor vessel shall be as described in Table 4.1.1 of the SAR: The applicable design codes shall be as described in Table 4.1.1 of the SAR.

5.4 Containment

A. The principal design parameters and applicable design codes for the primary containment shall be as given in Table 5.2.1 of the SAR.

B. The secondary containment shall be as described in Section 5.3.2 of the SAR and the applicable codes shall be as described in Section 12.1.1.3 of the SAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.2 of the SAR.

5.5 Fuel Storage

A. The new fuel storage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

5.6 Seismic Design

The reactor building and all contained engineered safeguards are design for the maximum credible earthquake ground motion with an acceleration of 20 percent of gravity. Dynamic analysis was used to determine the earthquake acceleration, applicable to the various elevations in the reactor building.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 74 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NO.3

DOCKET NO. 50-249

1.0 INTRODUCTION

By letters dated July 18, 1983 (Ref. 1) and August 25, 1983 as supplemented by letters dated November 3, 10, and 30, 1983, two letters dated December 13, 1983, and a letter dated December 16, 1983 Commonwealth Edison Company (CECo) (the licensee) proposed to amend Appendix A of Facility Operating License No. DPR-25. The requested amendment furnished information to support authorization for Dresden 3 to install, in place of eight standard control blades, eight lead control blades designed and built by ASEA-Atom and to support Cycle 9 operation of Dresden 3 with reload fuel supplied by and the associated analyses performed by Exxon Nuclear Company.

The ASEA-Atom blades are being tested as part of a demonstration program sponsored by EPRI aimed at qualifying a new blade design which would provide a greater exposure lifetime than the current design. In support of their proposal, the licensee has submitted a Technical Report TR-BR 82-98, Revision 1 (Ref. 2) for review.

The Dresden 3 Cycle 9 (D3C9) reload will consist of 408 fuel bundles fabricated by Exxon Nuclear Company (ENC). These 8x8 bundles are comprised of 63 active fuel rods and one inert water rod. During Cycle 9 operation the ENC fuel will reside with the 316 General Electric (GE) fuel assemblies presently in the core. In support of the D3C9 reload Commonwealth Edison Company (CECo) submitted topical reports which described the steady-state reload analysis, XN-NF-83-47, the plant transient analysis, XN-NF-83-58, and the loss-of-coolant accident (LOCA) analysis, XN-NF-81-75, Supplement 1.

Notices of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested actions in the July 18 and August 25, 1983 letters were published in the Federal Register on November 22, 1983 (48 FR 52807 and 52808). No request for hearing was received. A verbal comment from Mr. R. Minue of the Illinois Department of Nuclear Safety was received on December 5, 1983. His concern was related to the indications of cracking in ASEA-Atom control blades at high burnup as discussed in the licensee's November 10, 1983 supplemental letter. The blade cracking issue is addressed in Section 2.5 of this safety evaluation. The supplementary letters furnished clarifying information needed by the staff but made no changes in the content of the amendments and were, therefore, encompassed within the prenotices published November 22, 1983.

2.0 EVALUATION OF THE USE OF THE ASEA-ATOM CONTROL BLADES

2.1 GENERAL DESCRIPTION OF BLADE DESIGN AND PRESENT OPERATING SEQUENCE

The ASEA-ATOM (A-A) blades to be installed in Dresden Unit 3 have been designed to be mechanically compatible with existing blades. The blade profile is quite close to the standard blade and the velocity limiter and drive coupling portions are identical. The blades may be manipulated with the same handling tools as used on the standard blade. The blade weight is slightly less than the standard blade. The absorber material is vibratory-compacted B_4C but the blade design permits significantly more boron to be placed in the blade.

Dresden Unit 3 is currently operating with the Exxon single sequence control strategy. This means that the same control rods remain in the core throughout the Cycle (as opposed to periodic sequence changes in previous cycles). The A-A rods will be among those remaining in the core in order to maximize their exposure.

2.2 MATERIALS COMPATIBILITY

There will be two types of A-A blades used in D3C9. Four of the eight blades will have only B_4C as an absorber material, and four will have both B_4C and hafnium metal as absorber materials. The hafnium will comprise only the top six inches of the absorber section of these four blades. This design provision has been made to allow additional blade lifetime and reduce internal pressure in the blades. The use of hafnium in control blades has previously been approved for GE test blades in Peach Bottom, and is an alternative for the silver-indium-cadmium (Ag-In-Cd) used in Westinghouse reactors. The staff is unaware of any materials problems associated with the use of hafnium, and finds this aspect of the design acceptable.

The absorber in the A-A blade design is contained in horizontally drilled absorber holes in low-carbon stainless steel sheets. The staff's review of the mechanical design of the blades included a request for additional information (Ref. 3) related to the potential for blocking of the individual slits which interconnect these holes to equalize internal gas pressure in each blade wing. The applicant's response (Ref. 4) provides adequate assurance that there is no potential mechanism for blocking gas communication between the B_4C holes.

In addition, the staff evaluated additional information furnished by Commonwealth Edison (Ref. 4) on the conservatism of a 10 percent helium release rate (from B_4C) on blade temperature calculations, maximum internal gas pressure, mechanical strength and strain design requirements, use of gridpads, and the seismic design. Commonwealth Edison (Ref. 4) provided justification that each of these concerns has been addressed satisfactorily in the design of the control blades.

2.3 NUCLEAR DESIGN CHARACTERISTICS

The nuclear design characteristics of the improved A-A control blades has been performed by A-A with the PHOENIX lattice and depletion code. While this code has not been reviewed by the staff, a sufficient description of it has been included (Ref 2) to permit the conclusion that it is acceptable for use in performing the comparsions between the neutronic characteristics of the standard and A-A blades that are presented.

The code has been used to compare reactivity worths at cold xenon-free conditions and hot voided and unvoided conditions as a function of fuel burnup. In addition power distribution effects and absorber depletion effects have been studied. The conclusions of the analyses are discussed below.

The presence of a larger boron inventory in the rods implies a greater reactivity worth. The calculations by A-A have shown that the worth of the all B_4C rods is 6 to 9 percent greater than that of the standard rods. A control blade containing all hafnium would have about the same worth as the standard blade.

An important effect of the increased rod worth is to increase the shutdown margin. However, the increase in shutdown margin will be small for Dresden 3 since there are only eight of the stronger control rods and they will be placed in low worth regions of the core. Another effect of the increased boron content is a steeper flux gradient in assemblies surrounding an inserted control blade. The maximum difference is in the wide-wide corner and is about 5 percent. The difference at the LPRM location is only about 0.5 percent. These differences are accounted for in the reload analyses. The increased blade worth may cause the consequences of a rod withdrawal or rod drop event to be more severe. The effect of the presence of the A-A rods in Dresden 3 will be addressed for each reload containing them.

The increased boron loading of the blades also provides a longer exposure lifetime. A-A calculations show that the improved blade will have a 60 percent greater life if end-of-life is defined as a 10 percent reduction in blade worth. If the lifetime is determined on the basis of equal end-of-life worths, the improved rod would have more than twice the lifetime of the standard rod.

2.4 CONTROL ROD MANEUVERING

The A-A control rods are essentially identical in exterior envelope to the standard rods. The all B_4C rods are about 12 pounds lighter than current rods and the rods with hafnium tips are about 7 pounds lighter. Thus, the insertion speed should be greater for these rods. However, the presence of friction pads rather than rollers and an open central structure (increasing flow resistence) tends to offset the smaller weight. It is concluded that the insertion (scram) speed will not be significantly affected by the improved rods. The scram speed will be measured as part of the startup testing program.

2.5 BLADE SURVEILLANCE PROGRAM

By letter dated November 10, 1983 (Ref. 5) the licensee informed the staff that evidence of cracking with some loss of B_4C had occurred in similar rods being used in a Swedish reactor. Based on the proposed positioning of the eight lead A-A rods in the Dresden 3 core, the burn-up of the rods in the Swedish reactor, at the time the cracking was discovered, was greater than that which will occur during two 18 month cycles in Dresden 3. However, the lead rod burnup will be greater after three 18 month cycles than the burnup of the rods in the Swedish reactor.

Despite this, the staff has concluded that, because there are differences between the two sets of rods, concerns relating to their use are alleviated. In addition, the licensee has proposed an extensive monitoring program while they are being used at Dresden 3 so that indications of inferior performance will be detected promptly. These factors are significant enough for the staff to conclude that the Swedish problems would not be expected at Dresden 3. First, the stainless steel in the rods to be used in Dresden 3 has been fabricated with tighter chemistry control than that used in the blades used in the Swedish reactor. Second, nondestructive examination of the Dresden 3 A-A rods will be conducted following each usage cycle. Tests will be performed to check dimensional stability, corrosion effects and the integrity of the B_4C containment. A high resolution TV camera will be used for visual inspection, a guaging fixture for dimensional stability and a neutron transmission measurement for demonstrating B_4C presence. After the third 18 months cycle, an extensive examination of one or more rods will be made after their removal from the core.

Based on the above and upon the fact that four of the rods use hafnium instead of B_4C in the top six inches making them less susceptible to IGSCC from B_4C swelling, the staff has concluded that there is not a cracking-related safety concern from use of A-A rods in Dresden 3.

2.6 SUMMARY

On the basis of its review the staff has concluded that the use of the A-A improved control blades in Dresden 3 is acceptable. This conclusion is based on the following considerations:

1. The improved blades are mechanically and hydraulically compatible with the present control blades.

2. Only eight of the rods will be installed in the reactor.

3. The nuclear characteristics of the blades have been determined by acceptable methods.

4. The presence of the blades will be taken into account in the design and analysis of core reloads.

5. Sufficient experience has been had with the rod design in other (Swedish and Finnish) BWRs to permit the conclusion that they will operate without significant deterioration.

6. A satisfactory surveillance program has been established to monitor the blade performance.

3.0 EVALUATION OF THE DRESDEN 3 CYCLE 9 RELOAD SUBMITTALS

3.1 BACKGROUND

The D3C9 core will consist of 184 fresh ENC XN-2 8x8 fuel assemblies, 224 once-irradiated ENC XN-1 8x8 fuel and 316 GE 8x8 fuel assemblies. The ENC XN-2 8x8 fuel design is described in the approved generic report on the jet-pump (JP) BWR fuel design (XN-NF-81-21). This design is acceptable for use in the D3C9 reload with the exception of a previously applied burnup restriction on MAPLHGR limit (see section 3.2 of this report) and several conditions of approval on the generic fuel design report. These conditions are:

(1) The licensee must confirm that the design power profile shown in Fig. 5.10 of XN-NF-81-21 bounds the power limits for the application in question.

(2) Unless RODEX2 (XN-NF- 81-58) is approved without modification, the licensee must confirm or redo the following analyses, which were reviewed on the basis of RODEX2 results: design strain, external corrosion, rod pressure, overheating of fuel pellets, and pellet cladding interaction.

(3) Until such time that the Exxon revised cladding swelling and rupture models (XN-NF-82-07) are approved and incorporated in the ENC ECCS evaluation model, a supplemental calculation using the NUREG-0630 cladding models must be provided on a plant-specific basis each time a new ECCS analysis is performed.

(4) The licensee must make sure that the fuel performance code that is used to initialize Chapter 15 accident analyses has current NRC approval.

The staff has evaluated these four conditions during the course of our review, and its conclusions are described in the following paragraphs.

3.1.1 Power History

The licensee stated in the D3C9 reload submittal (XN-NF-83-47) (Ref. 6) that the D3C9 expected power history is bounded by the design profile in Fig. 5.10 of XN-NF-81-21 (Ref. 7). The staff has reviewed the references relating to the power history and concludes that the Cycle 9 power history is within the design limit and condition 1 is satisfied.

3.1.2 RODEX2 -- Strain, Corrosion, Rod Pressure, Overheating of Fuel Pellets, and Pellet - Clad Interaction (PCI) Analyses

The analyses of strain, corrosion, rod pressure, overheating of fuel pellets, and PCI were described in the approved JP-BWR fuel design. The staff has completed the review of the RODEX2 code used in this analysis and approved it with some modifications for licensing applications. Using the approved version limits on these physical parameters would not be exceeded throughout the entire lifetime. Since these analyses bound the Cycle 9 applications, the staff concludes that these analyses are acceptable for Cycle 9.

3.1.3 Cladding Swelling and Rupture

The cladding swelling and rupture models in XN-NF-82-07 (Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model) have been approved for use in the ENC ECCS evaluation model and have been incorporated in the approved ENC EXEM/BWR ECCS model. Since ENC used that approved swelling and rupture model for cladding in ECCS analysis, Condition 3 has been satisfied.

3.1.4 LOCA Initial Conditions

ENC used the recently approved steady-state code, RODEX2 (XN-NF-81-58), to calculate Cycle 9 LOCA initial conditions including stored energy and rod pressure for the ENC EXEM/BWR evaluation model. Thus Condition 4 is satisfied by the use of the approved code RODEX2.

3.2 MAPLHGR LIMIT

The MAPLHGR limit for XN-1 fuel during Cycle 8 operation was approved for burnups only up to 10,000 MWd/MTU due to the use of then unapproved code RODEX2. Subsequently the licensee requested that the 10,000 MWd/MTU limit be extended to 15,000 MWd/MTU, which was also approved (Ref. 8). Since the staff has approved the RODEX2 code, the licensee has confirmed that the MAPLHGR limit remains the same with the use of the approved RODEX2 code (Ref. 9). The staff finds this acceptable.

The MAPLHGR limit for XN-2 fuel in Cycle 9 is the same as the one for XN-1 fuel because of identical fuel design. The staff concludes that the MAPLHGR limit is acceptable for XN-2 fuel assemblies in Cycle 9.

3.3 SUMMARY

The NRC staff has reviewed the Dresden 3 fuel design and analyses for the Cycle 9 reload, and concludes that they are acceptable for Cycle 9 operation.

4.0 NUCLEAR DESIGN

The nuclear design of the Cycle 9 reload has been performed in accordance with the procedures described in XN- NF-80-19. The procedures have been previously used and approved for this purpose (see, for example, Dresden 3 Cycle 8 reload) and their use for Cycle 9 is acceptable. The results of the design analyses are given in Section 4.0 of XN-NF-83-47 (Ref. 6) including Table 4.1 and in Table 3.2 of XN-NF-83-58 (Ref. 6). These results are within the range normally expected for BWR reloads and are acceptable.

The use of eight A-A control blades for Cycle 9 has been approved as discussed in Section 2 of this Safety Evaluation.

4.1 TRANSIENT AND ACCIDENT ANALYSIS

The control rod withdrawal error, the fuel loading error and the rod drop accident were evaluated for Cycle 9. The use of the Single Sequence Control strategy (in which rods inserted during power operation have low worth) assures that the control rod withdrawal error will not be limiting. Using a Rod Block Monitor setting of 110 percent of full power results in a ΔCPR of only 0.16. The maximum change in CPR due to a fuel loading error is 0.19 and this event is not limiting either.

The control rod drop accident evaluation yields a value of 85 calories per gram for the maximum deposited fuel enthalpy. This is well below the staff's acceptance criterion of 280 calories per gram.

The effect of the presence of the eight A-A control blades on rod withdrawal and rod drop events has been considered by the licensee. The A-A blades will be located in low reactivity positions within the core and thus will not be the limiting rods for the rod withdrawal event. The startup withdrawal sequences were examined and the maximum potential ejected rod worth for the A-A blades was likewise found to be below that for the standard blades. THe resultant peak enthalpy was also lower for these blades. The staff concludes that the presence of the A-A blades has been adequately evaluated.

5.0 THERMAL-HYDRAULIC DESIGN

5.1 BACKGROUND

The review of the thermal hydraulic aspect of D3C9 consisted of the following:

(a) the operating safety limit minimum critical power ratio (OLMCPR),
(b) thermal-hydraulic stability,
(c) the Technical Specification changes.

The objective of the review was to confirm that the thermal-hydraulic design of the reload core was accomplished using acceptable analytical methods, to confirm that an acceptable margin of safety from conditions which would lead to fuel damage during normal operation and anticipated operational occurrences (AOOs) is provided, and to confirm that the Cycle 9 core is not susceptible to thermal-hydraulic instability.

5.2 MINIMUM AND OPERATING LIMIT CPR

The methodology for determining uncertainties and their application in determining the MCPR limit is contained in XN-NF-80-19 Volume 1 (Ref. 10) and XN-NF-512 (Ref. 11) and XN-NF-524 (Ref. 12). XN-NF-524 Volume 1 has been reviewed and approved by the staff. (Ref. 13)

The staff has completed the generic review of XN-NF-512 (Ref. 14) and XN-NF-524 (Ref. 15) and has concluded that the methodology for applying the XN-3 mean and standard deviation to arrive at a 1.05 for ENC fuel and a 1.06 for GE 8x8R fuel is acceptable.

Various operational transients could reduce the MCPR below the intended safety limit. The most limiting transients have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (ΔCPR). Table 2.1 of XN-NF-83-58 contains the results of these analyses. The transient which resulted in the largest ΔCPR was the load rejection without bypass.

The ΔCPR for the load rejection without bypass was calculated using the statistical methodology described in XN-NF-81-22 (Ref. 16), which has been reviewed and approved by the staff (Ref. 17). Based on this analysis the applicant has proposed a ΔCPR of 0.25 at a 95% probability level. The addition of this ΔCPR to the safety limit MCPR results in an operating limit MCPR (OLMCPR) of 1.30 for the ENC and GE 8x8 fuel designs, and 1.31 for the GE 8x8R design (Ref. 6).

Until the staff completes its generic review of XN-NF-79-71 (Ref. 18) the staff will require that code uncertainties be accounted for using the methods discussed in the safety evaluation report on the GE ODYN code (Ref. 19) as described for implementation in the staff safety evaluation for Dresden 3 Cycle 8 (Ref. 20).

Such a procedure for Dresden 3 requires that an ENC code uncertainty value of 0.022 ΔCPR/ICPR be applied deterministically to ΔCPR calculations. When this Δ CPR is added to the MCPRs the resultant OLMPCRs are 1.33 for ENC and GE 8x8 fuel designs and 1.34 for GE 8x8R fuel (Ref. 6).

The staff concludes that such an increase in ΔCPR acceptably bounds the code uncertainties and that the limits so derived will assure that the safety limit MCPR is not violated in the event of any anticipated transients.

5.3 THERMAL-HYDRAULIC STABILITY

The thermal-hydraulic stability of the Cycle 9 core was analyzed using the methods described in XN-NF-80-19, Volume 1, Supplement 2. The calculated decay ratio at the natural circulation - 100% rod line intersection (which is the least stable physically attainable point of operation) is 0.33. The calculated decay ratio for Cycle 8 was 0.45. The smaller decay ratio reported for Cycle 9 operation is attributed to the use of higher inlet orifice loss coefficients (which are more representative of the hydraulic characteristics of the ENC fuel assembly) in the Cycle 9 core stability analysis. Based on the fact that jet pump BWRs are not permitted to operate in the natural circulation mode and the fact that the decay ratio shows a large margin of stability, the staff concludes that the stability analysis of the Cycle 9 core is acceptable.

5.4 TECHNICAL SPECIFICATIONS

The licensee has submitted proposed Technical Specifications for D3C9 operation (Ref. 6). Section 3.5.K specifies the operating limit MCPRs, which are 1.33 for the ENC and GE 8x8 fuel designs and 1.34 for GE 8x8R fuel when scram times are less than or equal to 2.58 seconds. When the measured scram time becomes greater than 2.58 seconds the OLMCPR must be increased using the equation specified in Appendix A to XN-NF-83-47. Both the OLMCPR limit and the equation for adjusting the OLMCPR are currently in the Dresden 3 Technical Specifications. The only change to Technical Specification 3.4.K is to revise Figure 3.5-2 to incorporate the ENC curves for determining the OLMCPR for core flows less than rated flow. The revised Figure 3.5-2 is determined using the ENC methods documented in XN-NF-81-84, which is still under review by the staff. However, the review has progressed to the point where the staff concluded that the ENC methodology and the calculated results are acceptable for the D3C9 reload. The staff,therefore, has concluded that the revised Figure 3.5-2 is acceptable.

5.5 FINDINGS

The staff has reviewed the thermal-hydraulic design for the D3C9 reload core and has found that the results of analyses (XN-NF-83-47) support the proposed operating limit MCPRs, which avoid violation of the safety limit MCPR for design transients. The staff, therefore, concludes that this core reload will not adversely affect the capability to operate Dresden 3 safely during Cycle 9 operation and the proposed Technical Specification 3.5.K and the revised Figure 3.5-2 of the Technical Specifications discussed above are acceptable.

6.0 CONCLUDING SUMMARY

The staff has completed its review of D3C9 submittals including XN-NF-81-75 Supplement 1, XN-NF-83-47, XN-NF-83-58 and information relating to the use of eight ASEA-Atom control blades and found that they are acceptable. The staff thus concludes that Cycle 9 operation for Dresden 3 with the eight ASEA-Atom control blades and with 184 fresh ENC XN-2 fuel assemblies is acceptable.

7.0 ENVIRONMENTAL QUALIFICATION

The staff has determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, the staff further concludes that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR 51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal, need not be prepared in connection with the issuance of this amendment.

8.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

9.0 ACKNOWLEDGEMENT

The following staff members have contributed to this evaluation:

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R. Gilbert

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REFERENCES

1. Letter to H. R. Denton, NRC, from B. Rybak, Commonwealth Edison, dated July 18, 1983.

2. Technical Report TR-BR 82-98, Revision 1, "Performance Verification of an Improved BWR Control Blade Design" dated May 5, 1983.

3. Letter to D. L. Farrar, Commonwealth Edison, from D. Crutchfield dated November 1, 1983

4. Letter to H. R. Denton, NRC, from B. Rybak, Commonwealth Edison, dated November 30, 1983.

5. Letter to H. R. Denton, NRC, from B. Rybak, Commonwealth Edison, dated November 10, 1983, with attachment.

6. Letter to H. R. Denton, NRC, with attachments, including XN-NF-83-47, XN-NF-83-58 and XN-NF-81-75, Supplement 1, from B. Rybak, Commonwealth Edison, dated April 25, 1983.

7. XN-NF-81-27 "Generic Design Report - Mechanical Design for Exxon Nuclear Jet Pump BWR Fuel Assemblies," October, 1981.

8. Letter to D. Farrar, Commonwealth Edison from D. Crutchfield dated July 7, 1983.

9. Letter to H. R. Denton from B. Rybak, Commonwealth Edison, dated December 13, 1983.

10. XN-NF-80-19 (p), Volume 1 and Supplements 1 and 2, May 1980 Exxon Nuclear Methodology for Boiling Water Reactor Neutronics Methods for Design and Analysis.

11. XN-NF-512(P), Revision 1, The XN-3 Critical Power Correlation, March 1981.

12. XN-NF-524(P), Exxon Nuclear Critical Power Methodology for Boiling Water Reactors, November 1979.

13. Letter to G. Owsley, Exxon Nuclear Company from J. Miller dated April 7, 1982.

14. Letter to G. Owsley, Exxon Nuclear Company from H. Bernard dated July 22, 1982.

15. Letter to J. C. Chandler, Exxon Nuclear Company from C. Thomas dated October 31, 1983.

16. XN-NF-81-22(P) Generic Statistical Uncertainty Analysis Methodology, September 1981.

17. Letter to J. C. Chandler, Exxon Nuclear Company from C. Thomas dated October 28, 1983.

18. XN-NF-79-71(P), Revision 2, Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, November 1981.

19. Letter to Dr. G. G. Sherwood, General Electric Company from R. L. Tedsco dated February 4, 1981.

20. Letter to L. Del George, Commonwealth Edison Company, from J. Hegner dated April 29, 1982.